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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555
Revised: May 26, 1982

SCHEDULE AND OUTLINE FOR DISCUSSION
266TH ACRS MEETING
June 3-5, 1982
WASHINGTON, DC

Thursday, June 3, 1982, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 8:45 A.M. — ACRS Chairman's Report (Open)
- 1.1) Opening Statement
- 1.2) Items of interest regarding ACRS activities

2) 8:45 A.M. - 12:45 P.M. — Reactor Pressure Vessel Integrity (Open)
- 2.1) 8:45 A.M.-9:15 A.M.: Report of ACRS Subcommittee on Metal Components (MB/EI)
- 2.2) 9:15 A.M.-12:45 P.M.: Meeting with NRC Staff and representatives of the nuclear industry

12:45 P.M. - 1:45 P.M. — LUNCH

3) 1:45 P.M. - 3:45 P.M. — Quantitative Safety Goals (Open)
- 3.1) Discuss proposed ACRS report to NRC regarding NUREG-0880, Safety Goals for Nuclear Power Plants: A Discussion Paper (DO/JMG/GRQ)

4) 3:45 P.M. - 5:45 P.M. — Reactor Safety Research (Open)
- 4.1) Discuss proposed ACRS report to NRC regarding the proposed NRC Safety Research Budget for FY 1984-85 and the long-range aspects of the "out-years" (1986-88) (CPS/et al/SD)

Portions of this session will be closed as necessary to discuss information the premature release of which would be likely to significantly frustrate the performance of the Committee's statutory function.

5) 5:45 P.M. - 6:15 P.M.	5.1) Report by C. P. Siess regarding 5/18/82 hearing on NRC Safety Research Program (CPS/SD)

Page Revised

Friday, June 4, 1982, Room 1046, 1717 H Street, NW, Washington, DC

6) 8:30 A.M. - 12:30 P.M.	Midland Plant Units 1 and 2 (Open) 6.1) 8:30 A.M.-9:00 A.M.: Report of ACRS Subcommittee (DO/DCF) 6.2) 9:00 A.M.-12:30 P.M.: Meeting with NRC Staff and Applicant Portions of this session will be closed as necessary to discuss Proprietary Information related to this matter.
12:30 P.M. - 1:30 P.M.	LUNCH
7) 1:30 P.M. - 2:00 P.M.	Discuss Items for Meeting with NRC Commissioners (Open) 7.1) Disuss the following topics for the meeting with the NRC Commissioners . "Thermal Shock" of Reactor Pressure Vessels - The ACRS Acting Subcommittee Chairman (M. Bender) will provide a brief status report of activities related to the Committee's review and evaluation of the proposed NRC Staff plan of action to resolve this issue (see memo from NRC Chairman Palladino to Dr.P.G. Shewmon, ACRS Chairman, dated 3/25/82) . Quantitative Safety Goals - The ACRS Subcommittee Chairman (D. Okrent) will provide a status report regarding activities related to the ACRS review and development of comments regarding NUREG-0880, Safety Goals for Nuclear Power Plants, A Discussion Paper, dated 2/82 . Proposed ACRS Review of the Clinch River Breeder Reactor - The ACRS Subcommittee Chairman (M.W.Carbon) will make a brief presentation regarding the anticipated scope of and schedule for ACRS review of the CRBR

- Reactor Pressure Vessel Liquid Level/Inventory Instrumentation - The ACRS Subcommittee Chairman (William Kerr) will present a brief summary of the Committee's report dated 4/6/82 regarding this topic. Members of the Committee will be prepared to respond to questions regarding the Committee's comments and recommendations
- Proposed NRC Long-Range Research Program Plan - The ACRS Subcommittee Chairman (C.P. Siess) will present a brief summary of the Committee's report dated 4/5/82 regarding the Draft NRC Long-Range Research Plan for FY 1984-88 dtd. 3/15/82. Members of the Committee will be prepared to respond to questions the Commissioners may have regarding this matter.

8) 2:00 P.M. - 3:30 P.M. Meeting with NRC Commissioners (Open)
   8.1) Meeting with NRC Commissioners to discuss items noted above

9) 3:30 P.M. - 4:45 P.M. Future ACRS Activities
   9.1) Anticipated Subcommittee activities (PGS/MWL)
   9.2) Proposed ACRS activities (PGS/RFF)
   9.3) 3:45 P.M.-4:15 P.M.: Report by Dr. Moeller regarding consideration of seismic events in emergency planning (DWM/HA)
   9.4) 4:15 P.M.-4:45 P.M.: Report by Dr. Moeller regarding control room habitability in nuclear plants (DWM/HA)

10) 4:45 P.M. - 6:30 P.M. Quantitative Safety Goals (Open)
   10.1) Discuss proposed ACRS report to NRC regarding NUREG-0880, Safety Goals for Nuclear Power Plants, A Discussion Paper (DO/JMG/GRQ)

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Saturday, June 5, 1982, Room 1046, 1717 H Street, NW, Washington, DC

11) 8:30 A.M. - 12:30 P.M.

Preparation of ACRS Reports (Open/Closed)

11.1) Discuss proposed ACRS reports to NRC regarding:

- 11.1-1) 8:30 A.M.-9:30 A.M.: Thermal Shock of Reactor Pressure Vessels (MB/EI)
- 11.1-2) 9:30 A.M. - 10:30 A.M.: Midland Plant Units 1 & 2 (DO/DCF)
- 11.1-3) 10:30 A.M.-12:30 P.M.: Quantitative Safety Goals (DO/JMG/GRQ)

Portions of this session will be closed as necessary to discuss Proprietary Information and information that will be involved in an adjudicatory proceeding.

12:30 P.M. - 1:30 P.M.

LUNCH

12) 1:30 P.M. - 3:30 P.M.

Complete ACRS reports to NRC (Open)

12.1) Discuss proposed reply to Commissioner Gilinsky's inquiry regarding seismic methodology proposed by Dr. P. Jennings (DO/RS)

12.2) Complete discussion of reports noted above

Portions of this session will be closed as necessary to discuss Proprietary Information and information that will be involved in an adjudicatory proceeding.

proposed rule will modify 10 CFR 50.34 (contents of applications; technical information) and contains the basic requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements".

**Metal Components*, June 7, 1982, Palo Alto, CA. The Subcommittee will be given a status report by nuclear reactor Steam Generator Owners Group on research results and any changes made in steam generator design/operation. Three Mile Island Unit 1 steam generator problems will also be discussed.

**Waste Management*, June 8, 1982, Washington, DC. The Subcommittee will review and comment on the Department of Energy's Public Draft of the National Plan for Siting High-Level Waste Repositories and Environmental Assessment; provide input for the Waste Management Chapter of the FY 1984 and FY 1985 Safety Research Program Review; review NRC Staff waste management activities; and discuss advances in waste management practices.

**Emergency Core Cooling Systems (ECCS)*, June 16 and 17, 1982, Idaho Falls, ID. The Subcommittee will discuss General Electric Company's request for a change in 10 CFR Part 100, Appendix K requirements, the NRC Staff's code audit capability, the LOFT ATWS test Nuclear Steam Supply System vendor code predictions and results, and the NRC work on operator accident guidelines and procedures.

**Reactor Radiological Effects*, June 23, 1982, Washington, DC. The Subcommittee will discuss NRC Staff proposed revision to 10 CFR 20 and the use of potassium iodide for thyroid blocking in the event of a radiation accident.

**Washington Public Power Supply System Unit 2 (WPPSS)*, June 23 and 24, 1982, Hanford, WA. The Subcommittee will continue the review of the application of Washington Public Power Supply System for an operating license for the WPPSS Nuclear Project Unit 2.

**Clinch River Breeder Reactor (CRBR) and Site Suitability*, June 24 and 25, 1982, Washington, DC. The Subcommittee will continue the site suitability review for the Clinch River Breeder Reactor.

**Perry Nuclear Power Plant Units 1 and 2*, June 28 and 29, 1982, Cleveland, OH. The Subcommittee will continue the review of the application of Cleveland Electric Illuminating Company for an operating license for the Perry Nuclear Power Plant Units 1 and 2.

**Systematic Evaluation Program*, June 30, 1982, Washington, DC. The Subcommittee will review the completion of the Systematic Evaluation Program review on Ginna.

**Grand Gulf Unit 1*, July 1, 1982, Washington, DC. The Subcommittee will continue the review of the Mississippi Power Company application for an operating license for Grand Gulf Unit 1.

**Extreme External Phenomena*, July 1, 1982, Washington, DC. The Subcommittee will review the Office of Nuclear Regulatory Research proposed FY 1984 and FY 1985 research funding and programs in this area for the Long-Range Research Plan.

**Reliability and Probabilistic Assessment*, July 1, 1982, Washington, DC. The Subcommittee will review the Office of Nuclear Regulatory Research proposed FY 1984 and FY 1985 research funding and programs for the Systems and Reliability Analysis (SARA) decision unit.

**Regulatory Activities*, July 6, 1982 (Tentative), Washington, DC. The Subcommittee will review proposed Regulatory Guides and Regulations.

**Safety Research Program*, July 7, 1982, Washington, DC. The Subcommittee will continue its review of the NRC Safety Research Program and budget for FY 1984 and FY 1985.

**Reactor Operations*, July 21 or 22, 1982, Washington, DC. The Subcommittee plans to discuss NRC's enforcement policy, the Inspection and Enforcement (IE) performance apprisal team inspection program and the IE regionalization program.

**Watts Bar*, Date to be determined (July), Washington, DC. The Subcommittee will continue the review of the application of Tennessee Valley Authority for an operating license for the Watts Bar Nuclear Power Plant Units 1 and 2.

**Safety Research Program*, August 11, 1982, Washington, DC. The Subcommittee will provide early input to the RES Staff for their preparation of the Long-Range Research Plan for FY 1985 through FY 1989.

**Transportation of Radioactive Materials*, Date and location to be determined. The Subcommittee will continue its review of the adequacy of the NRC procedures for certifying packages for transporting radioactive materials.

**Metal Components*, Date to be determined, Washington, DC. The Subcommittee will continue the review of pressurized thermal shock.

ACRS Full Committee Meeting

June 3–5, 1982: Items are *tentatively* scheduled.

**A. Midland Nuclear Plant*—Operating license.

**B. Quantitative Safety Goals*—Proposed NRC policy regard Quantitative Safety Goals for Nuclear Power Plants (NUREG-0880).

**C. Reactor Safety Research*—Proposed NRC Safety Research budget for FY 1984 and FY 1985.

**D. Reactor Pressure Vessel Integrity*—Proposed NRC action plan to resolve concerns regarding repressurization of reactor pressure vessels following rapid cooldown transients.

**E. NRC Regulations*—Proposed NRC regulations regarding safety related matters including Application of TMI-2 Lessons Learned to Operating Reactors (10 CFR 50.34); Applicability of License Conditions and Technical Specifications in an Emergency (10 CFR 50.54/50.72); Accreditation of Testing Organizations (10 CFR 50.49(a)); and Evaluation of Alternate Decay Heat Removal Systems (Task Action Plan A-45).

**F. ACRS Subcommittee Activities*—Discuss the status of designated ACRS Subcommittee activities regarding safety related matters including consideration of seismic events in emergency planning; and proposed changes in seismic design methodology.

**G. Meeting with NRC Commissioners (Tentative)*—Discuss ACRS activities regarding quantitative safety goals for nuclear power plants, integrity of reactor pressure vessels, ACRS plans for review of the CRBR, the NRC Long-Range Research Program Plan, and instrumentation for detection of inadequate core cooling.

**H. Three Mile Island Nuclear Plant Unit No. 1*—Briefing regarding causes of and status of steam generator tube damage.

July 8–10, 1982: Agenda to be announced.

August 12–14, 1982: Agenda to be announced.

Dated: May 14, 1982.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 82-13681 Filed 5-18-82; 8:45 am]

BILLING CODE 7590-01-M

[Docket No. 50-313]

Arkansas Power & Light Co. (Arkansas Nuclear One, Unit 1); Exemption

I

The Arkansas Power and Light Company (the licensee) is the holder of Facility Operating License No. DPR-51, which authorizes operation of Arkansas Nuclear One, Unit No. 1. This license provides, among other things, that it is subject to all rules, regulations and

Issue Date:

October 8, 1982

CERTIFIED

MINUTES OF THE
266TH ACRS MEETING
JUNE 3-5, 1982
WASHINGTON, DC

The 266th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC was convened by Chairman P. Shewmon at 8:30 a.m., Thursday, June 3, 1982.

[Note: For a list of attendees, see Appendix I. D. A. Ward and M. S. Plesset were not present for the meeting. D. W. Moeller was unable to attend on Thursday.]

The Chairman noted the existence of the published agenda for this meeting, and identified the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act (FACA) and the Government in the Sunshine Act (GISA), Public Laws 92-463 and 94-409, respectively. He also noted that a transcript of some of the public portions of the meeting was being taken, and would be available in the NRC's Public Document Room at 1717 H St. N.W., Washington, DC.

[Note: Copies of the transcript taken at this meeting are also available for purchase from the Alderson Reporting Co., Inc., 400 Virginia Ave. S.W., Washington, DC 20024.]

I. Chairman's Report (Open to Public)

[Note: Raymond F. Fraley was the Designated Federal Employee for this portion of the meeting.]

Chairman Shewmon indicated that written statements had been received from Mrs. Mary Sinclair and Mrs. Barbara Stimeris related to the ACRS review of Midland 1 and 2. He noted that Mrs. Sinclair had requested time to make an oral statement to the Committee during the presentation on Midland. Chairman Shewmon also noted that Commissioner James K. Asseltine had assumed his duties of NRC Commissioner as of May 16, 1982 bringing the Commission to its full level of five commissioners. Also mentioned was the testimony given by C. P. Siess regarding the NRC Safety Research Program before the Subcommittee on Energy Research and Production of the U. S. House of Representatives Committee on Science and Technology. Chairman Shewmon mentioned a set of questions received by the Committee from the Staff of this Subcommittee on Energy Research and Production which will be discussed by the ACRS in Executive Session at the end of the day.

II. Operating License Review of Midland Plant Units 1 and 2 (Open to Public)

[Note: David C. Fischer was the Designated Federal Employee for this portion of the meeting.]

[W. Kerr did not participate in the review of the Midland Plant.]

A. Report of the ACRS Subcommittee

D. Okrent reviewed the history of the Midland license for the Committee He mentioned that the ACRS had done a particularly detailed review of Midland in 1969 and 1970 because the site was one that had a higher population density within three miles from the plant than did other proposed nuclear power stations. D. Okrent referred to a Committee letter dated November 18, 1976 which identified issues which should be considered in the OL review (see Appendix IV).

D. Okrent indicated that there are some special issues applicable to Midland. He called the Committee's attention to a history of quality control deficiencies at Midland during the construction period, including some problems with cadwelds, bolts, and soil settling, as well as cracking at the foundation of the diesel generator building. He suggested that the Committee pay special attention to specific issues that dealt with the quality question. D. Okrent brought up a question concerning the seismic design rereview, a question of liquefaction problems with soils under many of the safety related structures and a dewatering scheme being proposed by the Applicant. Other topics mentioned for discussion were questions regarding whether a high point vent on the reactor vessel should be provided, whether provisions should be made for instrumentation to detect inadequate core cooling, whether less than favorable experience with high strength bolts required an explanation.

D. Okrent pointed out that there were no major issues regarding fire protection. He indicated that the Applicant is proposing an extensive program to evaluate systems interactions, similar to that being done at Indian Point. Although the integrated control system was modified somewhat to accommodate the process tertiary steam system he did not see a need for extensive Committee attention to these modifications. D. Okrent did point out that the Committee should decide whether to pursue the issue of turbine missiles as a specific or generic issue with regard to Midland.

D. Okrent identified several other potential issues which might be discussed as part of the Committee's Operating License Review (see Appendix IV):

- High copper content in the welds of the Midland 1 reactor vessel
- The status of the ongoing probabilistic risk assessment at Midland
- Commitment by the Applicant to install a third auxiliary feedwater pump in the nonseismic, Category-1 turbine building
- B&W emergency operating procedures
- Industrial security
- Steam generator overfill protection.

C. P. Siess summarized the ad hoc Subcommittee meeting on Midland Foundation Problems and Remedial Actions which was held on April 29, 1982. The problem at Midland is inadequately compacted fill that is partly granular and partly cohesive soils. He indicated that the consequence of this inadequate compaction was the differential settlement of certain safety related structures. This produced some cracking in the walls of some reinforced concrete structures. He indicated that the Subcommittee concluded, after presentations by the Staff and their consultants, that remedial measures being taken seemed appropriate to allay any particular concern about structural adequacy. He noted that the Subcommittee was satisfied with the dewatering system proposed by the Applicant to eliminate the hazard of liquefaction. However, the question of the seismic input to the liquefaction analysis was still open since the Subcommittee had not reviewed the seismic design spectrum during its meeting.

B. Statement by Mary Sinclair

Mary Sinclair, a citizen of Midland, Michigan, read a statement on the Midland Nuclear Plants (see Appendix V). M. Sinclair described the environment in the immediate vicinity of the Midland Plants, including the siting of an elementary school "immediately across the road from the Midland facility." She explained that her purpose was to present a public perception of the role of the Advisory Committee on Reactor Safeguards. The theme of her statement was that the public has lost confidence in the nuclear power plant licensing process.

C. Status of the NRC Staff Review

R. Hernan, NRR Project Manager for Midland, reviewed the SER open items individually, (see Appendix VI). A list of special review areas was presented as areas of particular concern to the NRC Staff. J. Ebersole questioned how the Staff was evaluating the soils settlement issue.

J. Kane, NRC Staff, indicated that the problem had been evaluated by measured building settlement and by making borings in the in-place fill material.

R. Hernan indicated that the Staff had looked closely at the unique process steam system at the Midland plant with regard to radiation monitoring in the case of a primary to secondary system leak. J. Ebersole pointed out that there is a vastly increased probability of secondary blowdown with such a system. B&W reactors are extremely sensitive to secondary system blowdown in view of the superheat design of the steam generators. He questioned whether the NRC Staff had looked into the combination of this increased probability of secondary blowdown in conjunction with a control system failure on feedwater overfilling the steam generator. This could result in an extremely rapid depressurization and thermal shock to the Midland 1 reactor vessel which does have a high copper content. He suggested that there is an unusual potential for very large thermal transients in this system. R. L. Tedesco, NRC Staff, pointed to the safety grade overfill protection system and the fact that only one steam generator would blow down should an accident occur. J. Ebersole expressed concern about the assumptions in the NRC analysis.

M. Bender expressed concern regarding the NRC's collective judgment as to the quality of the Midland plant. He questioned whether there was an integrated, comprehensive report on the problems of quality at Midland plant. R. L. Tedesco indicated that the Staff did not plan to produce an integrated report on this subject.

D. Okrent and R. Axtmann expressed concern about emergency preparedness and emergency planning at Midland. R. Axtmann inquired whether an emergency plan would be in place before startup. R. L. Tedesco indicated that a completed emergency plan might not be in place for low power operations, but that a tested plan must be available before the plant goes into full power operation.

R. Mattson, NRC Staff, indicated that steam generators should be protected against overfill from either the main or auxiliary feedwater systems. Equipment to provide this protection should be safety grade.

D. Okrent noted that this issue is particularly important on B&W plants because of their control sensitivity and he questioned the lack of urgency expressed by the NRC Staff at issuing a backfit requirement for operating plants. D. Okrent requested a written response within the next month regarding the NRC Staff position with respect to the issue of feedwater overfill protection. J. Ebersole requested that the NRC Staff include in its report an analysis of the consequences of continuing to pump cold main feedwater into the steam generator in the event of a main steam line failure. This procedure can lead to a severe secondary transient leading to the pressurized thermal shock problem in the reactor pressure vessel.

D. Quality Control Issues

W. Little, NRC Staff, Region III, presented a tabulation of NRC criteria for assessing contruction QA/QC at nuclear power plants (see Appendix VIII). As a result of the Staff's Systematic Assessment of Licensee Performance (SALP) review of Midland, the Staff had identified six areas which it plans to follow in more detail than currently required. M. Bender questioned how the Staff makes a final judgment regarding the overall plant adequacy. W. Little suggested that the Staff has to depend on its routine inspection program to assess the overall adequacy of plant construction.

J. Ebersole questioned how extensive the Staff effort would have to be in order to assure against a total failure of flow of service water. J. Kane, NRC Staff, indicated that the Staff has undertaken QA audit efforts sufficient to confirm loose fill, soft clays under pipes, the measurement of settlements and stresses on pipes. Based on an evaluation of the remedial measures taken by the Applicant, the Staff is convinced that the problems that have been identified are being adequately addressed.

D. Okrent requested an explanation of the six items that would require special quality assurance monitoring by the Staff. W. Little identified these as follows:

- Remedial actions related to soils problems
- Piping systems and supports
- Electrical power and supply distribution
- Instrumentation and control
- Design control and the control of design changes
- Reporting requirements and corrective action.

M. Bender suggested that the Staff prepare a comprehensive report identifying quality problems at the Midland site and containing an overall assessment of plant quality. R. Vollmer, NRC Staff, did not think the Staff would have any objection to preparing such a report. He indicated that, before this plant could be licensed, he expected Consumers Power to provide objective evidence that the plant had been designed and constructed in accordance with the application. M. Bender expressed concern that the construction problems found may suggest that greater care should have been taken during the construction phase. R. Vollmer thought that the audits the Staff is conducting regarding mechanical and structural details should provide sufficient assurance of the quality of construction.

E. Consumers Power Presentation Regarding Quality Assurance

D. W. Marguglio, Construction Quality Assurance Program Manager for Consumers Power Company (CPCo), described three major aspects of the quality assurance effort at the Midland site:

- NRC's increased Inspection Program
- External, independent audits and assessments by CPCo consultants (biennial audits)
- CPCo performed reinspections and rereviews.

The Committee discussed the apparent buildup in the quality assurance organization and its relationship to the fill material and electrical equipment qualification issues. P. G. Shewmon questioned whether the independent audits being conducted by Consumers Power have uncovered anything in the six areas that the NRC inspection teams have been concentrating their efforts. D. Marguglio indicated that a recent review found the timeliness of quality assurance corrective actions to be quite satisfactory.

J. Ebersole pointed out that numerous significant targets are in the direct path of potential turbine missiles. He questioned the position of the NRC Staff regarding the potential problem of both a turbine stop valve and control valve failure which could lead to turbine overspeed and disc failures. He mentioned attempts by the Applicant to put two trip systems on a single set of valves as a solution to the problem. R. Klecker, NRC Division of Engineering, explained the NRC's turbine missile guidelines as shown in Standard Review Plan III.5.1.3 (see Appendix X). He compared the Applicant's values for missile generation, strike and damage probabilities with NRC's Standard Review Plan numbers.

F. Seismic Review

J. Kimball, NRR Staff Seismologist, explained the Staff's position on the Midland Plant. Two alternatives were given to the Applicant after the Applicant's analysis at the construction permit stage had been reviewed and found to require reanalysis. The Applicant decided to use the site specific spectrum to replace the 0.12g modified Housner spectrum which was the original Midland design spectrum. The Staff and Applicant agree that the 84th percentile in the Midland site specific response spectra is a conservatve representation for the ground motion at the Midland site (see Appendix XI).

L. Reiter, Section Leader for Seismology in the NRR Division of Engineering, made some general comments regarding probabilistic estimates for the safe shutdown earthquake. He noted that reliance upon probabilistic estimates for very long return period earthquakes is not the way to alleviate concerns regarding earthquakes greater than the safe shutdown earthquake. In answer to a question by D. Okrent, L. Reiter indicated that one possible way to alleviate some of these concerns would be to make a study of events, the probability of which is high enough to be accurately estimated by available procedures, in order to develop a base from which to extrapolate less likely events. G. Knighton, NRC Staff, indicated that simply raising the g value for the plant site would not give confidence from the seismic point of view, as would a closer look at the capacity of the equipment or the design of the equipment to withstand more severe shaking. The Committee discussed the design of structures at nuclear plants in general with regard to their ability to withstand a seismic event.

R. Kennedy, President of Structural Mechanics Associates, consultant to Consumers Power, briefly summarized the criteria for the seismic margin review at the Midland plant (see Appendix XIII). R. Kennedy described the screening process to select structural elements, components, and distribution systems for seismic safety margin evaluation, and presented an example of analysis results for the borated water storage tank at Midland. There were no questions from the Committee.

T. R. Thiruvengadam, Consumers Power Co., reviewed the soils exploration program at Midland with regard to liquefaction potential and margins. He identified the diesel generating area, and the railroad bay area of the auxiliary building as the principal structures for which remedial measures against liquefaction were found necessary (see Appendix IX). He indicated that if these areas are dewatered and the ground water level is maintained at or below elevation 610, the structures would be safe against liquefaction for earthquakes with peak ground accelerations of 0.19g. He added that during normal operations of the dewatering system, the water level is maintained at elevation 595.

T. R. Thiruvengadam indicated that for an earthquake of magnitude 6 or 0.19g acceleration there is a factor of safety of 1.5 against the potential for liquefaction and for a 0.25g acceleration there is a factor of safety of 1.1 against the potential for liquefaction. In answer to a question by D. W. Moeller, T. R. Thiruvengadam indicated that the safety factor of 1 would imply the onset of liquefaction.

D. Okrent summarized the various views of the ACRS consultants with respect to the seismic area. R. Holt, Western Geophysical Corp., consultant for Consumers Power, attempted to clarify and reconcile Midland numbers with the numbers estimated by Drs. Trifunac and Pomeroy, ACRS consultants.

G. Inadequate Core Cooling Instrumentation and Reactor Vessel Head Vent

R. Mattson, NRC Staff, indicated that a reactor vessel head vent will be required of the Applicant before licensing. The Staff and the Applicant have continued to discuss the core exit thermocouples as a means of detecting inadequate core cooling. R. Mattson indicated that these thermocouples would be upgraded and operational prior to fuel load. He added that the hot-leg monitoring system proposed by the Applicant is inadequate and has to be upgraded to include a vessel head tap.

L. Gibson, Section Head for Safety and Analysis, Consumers Power Co., presented Consumers Power's position with regard to venting their B&W designed reactor coolant system. He indicated that Consumers Power is in agreement with B&W, that the proper way to provide venting for the B&W design is through the use of vents at the top of the hot leg and the top of the pressurizer. L. Gibson expressed Consumers Power's belief that a level indicator in the reactor head does not provide additional margin for the operator to respond to an inadequate core cooling event. He indicated that Midland procedures call for trip of the reactor coolant pumps on a loss of subcooling margin in order to avoid void formation in the pumps. D. Okrent noted that there was definitely a philosophical difference between the Staff and the Applicant. R. Mattson indicated that, regardless of the Committee's report, the Staff was prepared to go to the Hearing Board with its current position.

R. Mattson mentioned the Semiscale/MOD-V which will model the B&W reactor system. The discussion involved recent TRAC calculations made by Los Alamos involving the ability to maintain single phase natural circulation cooling in the B&W design for certain small break LOCAs. He referred to a letter from the NRC Staff to H. Meyers of the Udall Committee which explains the Los Alamos calculations (see Appendix XIV) and suggested that the TMI-2 Hearing Board might require additional information regarding this matter. In response to an inquiry by D. Okrent, R. Mattson indicated that this matter was an open or outstanding issue which will be addressed in a supplement to the SER.

H. Slager, Consumers Power, provided the summary of experience at the Midland plant regarding bolting. During routine testing of reactor vessel anchor bolts, several failed in a ductile manner. They were found to be much softer than anticipated because of improper heat treatment. Because of this experience Consumers Power Company initiated a hardness testing program for all special purpose bolts. H. Slager noted that this is a QA problem which involved more than just record keeping. He indicated that the hardness test program was eventually extended to cover other bolts and similar problems were found with steam generator anchor bolts, reactor coolant snubber anchor bolts, and pipewhip restraint bolts (see Appendix IX). H. Slager indicated that Paladine Engineering Services was hired to perform an independent analysis of the Midland Reactor Vessel anchor bolts and found that the cracking mechanism was initial stress corrosion cracking followed by complete failure due to the low fracture toughness.

H. Slager explained that in order to avoid further stress corrosion cracking, Consumers decided to lower the prestress on the anchor bolts from 92 ksi to 6 ksi and add upper lateral supports to take up some of the potential seismic loads carried by the reactor vessel anchor bolts in the original design. H. Etherington suggested that it is not good engineering practice to let the design load exceed the prestress load on these bolts. The prestress load should be at least equal to the design load. T. R. Thiruvengadam indicated that that was the original intent, but the lost stiffness was now being taken up through the upper lateral supports.

Chairman Shewmon inquired whether the Staff had made any progress evaluating the use of this ASTM specification that has resulted in the placement of unsatisfactory material at two plants so far. C. D. Sellers, NRC Staff, indicated that the Staff does not have anything other than a technical assistance contract at Brookhaven that would address this matter. An NRC position addressing anchor bolt preload, material selection, hardness, inspection at receipt, and inspection in service would be formulated from the results of the Brookhaven contract.

J. J. Ray asked several questions pertaining to a.c./d.c. electrical system reliability. B. Harshe, Consumers Power, answered these questions as follows:

- Analysis for stability of the grid assumed a single failure such as a breaker that did not operate, line problems coincident with the fault such that there was stability long enough for backup relaying or backup switching to take place.

- With regard to d.c. supply, batteries are oversized and should last for approximately 4 1/2 hours under full load conditions.

- Load shedding analyses to verify extension of the 4 1/2 hour battery lifetime in the event of a blackout have not been done yet.

- All Consumers Power Nuclear Plants have top priority for restoration of power in the event of a blackout.

- Consumers Power System has blackstart capability through the use of the hydro facility at Leadington, diesel generators, and gas turbines.

C. Mark pointed out the unfavorable orientation of the plant turbines and questioned the NRC Staff's procedures for determining strike and damage probabilities. P. G. Shewmon asked the NRC Staff to explain their general approach with regard to the turbine missile strike probability P_2, and the damage probability P_3.

D. W. Moeller questioned whether the Applicant had considered the Bullock Creek Elementary School in its emergency planning. W. Beckman, Consumers Power, indicated that there were actually two questions involved, the first involving the status of the Emergency Plan and the second with respect to the elementary school. He first indicated that the Midland County Emergency Plan has been reviewed by the State of Michigan. He pointed out that the school lies in Midland County and has an evacuation plan using buses. Information in answer to additional questions by D. W. Moeller concerning emergency planning are as follows:

- Saginaw and Bay County Emergency Plans will be submitted to FEMA for review.

- The Dow Chemical Co. and Consumers Power have reciprocal agreements regarding an accident at the nuclear power plant with plans for protecting the personnel and shutdown of certain facilities in the DOW Plant.

- Dow Chemical personnel participated in an emergency drill at the nuclear power plant, but Midland plant personnel do not yet participate in drills at the chemical plant.

D. Okrent expressed concern regarding the question of small break LOCAs and possible difficulties with natural circulation for B&W plants, and the Midland plant in particular. He explained that formation of a bubble at the top of the reactor coolant system hot leg would most certainly interfere with natural circulation if the reactor coolant pumps were tripped. He expressed displeasure with the fact that this item was not mentioned in the SER and that the Committee was given insufficient information to make a technical evaluation of it. He suggested that he would be more comfortable if the Committee did not go beyond a recommendation for 5% power operation prior to resolution of this issue.

In answer to a question by C. Mark, T. J. Sullivan, Consumers Power, indicated that the subject of control room habitability in the event of noxious gas release from Dow Chemical had been addressed. In answer to a question by J. Ebersole concerning the competency of the diesel generator building to handle a transformer failure and consequent fire, R. Burg, Bechtel Power Corp., indicated that they had looked at fire and also explosion with regard to the diesel generators and that the diesel generators can be controlled remotely from the main control panel for an indefinite period of time.

III. Reactor Vessel Integrity (Open to Public)

[Note: Elpidio Igne was the Designated Federal Employee for this portion of the meeting.]

A. Report of ACRS Subcommittee on Metal Components

M. Bender explained that the purpose of this meeting was to discuss pressure vessel integrity in response to a request by Chairman Palladino with regard to the short term program associated with the pressurized thermal shock issue and provide recommendations as appropriate to the Commission and the NRC Staff. He cited recommendations made to the Commission several months ago that suggested that the NRC Staff seek to familiarize themselves with the composition of the materials that are in the vessels in question, and that operating procedures to protect against thermal shock were in place in those plants. He mentioned that an active audit program was being conducted at the H. B. Robinson plant and some other plants where there is a comparable concern.

M. Bender introduced ACRS consultants M. Wechsler, Z. Zudans, I. Catton, G. Irwin, T. Theofanous, H. Kouts, and E. Abbott who were present at this meeting. He indicated that an ACRS working group had been formed and has addressed three separate issues.

. Thermal hydraulics questions that influence vessel temperature

. The materials question using the fracture mechanics approach

. Operational procedures.

M. Bender presented some of the problems that are posed with respect to this issue. He indicated that the presence of copper in the welds of the reactor vessel is the dominant problem that determines the amount of fracture toughness lost. It is important to judge the condition of the vessels. He added that a second problem involved the interpretation of fracture toughness determination based upon impact tests. Confusion in the interpretation of these hardness tests has influenced evaluation of how severe the thermal shock question really is. He mentioned another problem which involved understanding how control systems influence thermal transients and how the operator might respond if control systems do not work to control the thermal transient with existing control circuitry. It was mentioned that the question of whether the operator has a conflict in his operating decisions that prevents him from executing a timely, safe procedure is also important, especially regarding the adequacy of protection techniques.

M. Bender indicated that the NRC Staff has suggested that it would be worthwhile to reduce the fluence accumulation rate for vessels of concern. He suggested that the ACRS position would be to continue studying the problem, expecially with regard to proper control of operating conditions until the situation in better defined.

Chairman Shewmon suggested that the problem may be largely or completely avoided if the operator depressurizes the system to conditions near saturation. M. Bender mentioned an analysis by T. Theofanus of the way in which cooling rates could occur in the reactor vessel wall (see Appendix XV).

B. Presentation by the NRC Staff

H. Denton described the NRC approach to this problem as an action level or probability of pressurized thermal shock causing vessel failure in the range of 10^{-7} per vessel per year. F. Schroeder, NRC Staff, explained that the NRC approach involves trying to pick a limit on acceptable operation by analysing or constructing accident sequences and following the course of the events and probabilities that can be assigned to them. F. Schroeder discussed a tabulation of actual events and their characteristics (see Appendix XVI), including the final temperatures to which the water drops, and exponential time constant called beta to fit the actual transient with an exponential curve. F. Schroeder pointed to curves with values of temperature and pressure at which deterministic analyses predict cracks for a family of values of RT_{NDT}. He pointed out that such plots could actually define safe and nonsafe regions where a transient final temperature and specified beta can be plotted along with values of RT_{NDT} to identify the pressure necessary for crack initiation. From this procedure one could determine a pressure to stay below in order to avoid crack initiation. R. Klecker, NRC Staff, defined more specificially the NRC assumptions with regard to crack sizes including the length and depth of the crack and its location in the vessel. T. Schroeder pointed to cross plots which show the difference between the final temperature in the transient and the reference temperature in the vessel versus the probability of vessel failure for three different values of heat transfer coefficient. He pointed out that these curves are very steep and that a change in temperature of 20^{o} increased the estimated probability of failure, given a particular event, by orders of magnitude.

F. Schroeder suggested as an NRC criterion that the limit on RT_{NDT} for operation should be 230^{o} F for longitudinal welds. The Committee discussed the best estimate value of 230^{o} F and the uncertainties in estimates of probabilities of vessel failure in severe transients. F. Schroeder discussed possible actions for plants that do not currently meet the criteria. Mentioned were operations improvements, instrumentation improvements, and pressure limiting control systems. He indicated that credit would be given if full volumetric, nondestructive examination of the vessel was performed. He added that the ultimate solution for plants that would exceed such a criteria would be an annealing of the vessel.

F. Schroeder pointed to NRC Staff goals involving flux reduction considerations and defense indepth features dealing with limits on vessel material properties, upgrading operational procedures, and improved instrumentation to allow the operator a better chance to stay out of trouble or possibly hardware improvements in the form of automatic pressure controls, warming ECCS water and required flux reduction rate at some RT_{NDT} threshold.

Chairman Shewmon suggested that the Staff's approach had substantial conservatisms and he questioned whether the Staff had taken account of certain mitigating factors

- Operator actions to ameliorate the accident
- Warm prestress phenomena applied to the pressure vessel
- Probability of a crack size distribution
- Probability of crack initiation leading to a core melt.

H. Denton indicated that the Staff was using a bounding approach in order to produce a regulatory decision in as expeditious a manner as possible and avoid getting mired in the technical details.

C. Presentations by Representatives from the Nuclear Industry

1. Westinghouse Owners Group

D. Speyer, Chairman of the Analysis Subcommittee of the Westinghouse Owners Group (WOG) mentioned a WOG report submitted to the NRC on May 28 entitled, Summary of Evaluations Related to Reactor Vessel Integrity. He defined the objectives of the Owners Group program

- Demonstrate no near-term safety issues in Westinghouse plants
- Reveal generically developed methodologies and techniques for addressing pressurized thermal shock
- Provide input to economic deliberations by utility members (see Appendix XVII).

D. Speyer indicated that analysis has shown that decay heat is very important to analysis of pressurized thermal shock or reactor vessel decay heat transients. Small amounts of decay heat have a beneficial effect and the absence of decay heat will result in a more severe cooldown. P. G. Shewmon and W. Kerr expressed concern that Westinghouse was not considering operator misactions in its analysis of operator actions during a cooldown transient. D. Speyer indicated that although the worst operator action is considered the infinite time for response to terminate the event, the models that Westinghouse used for analysis do include incorrect or improper actions or misactions by the operator. D. Speyer indicated that the Westinghouse best estimate calculation showed for full system pressure approximately 290° F. Below this temperature one could potentially expect crack initiation.

After the Westinghouse approach was described including assumptions, M. Bender asked the NRC Staff whether it had evaluated the Westinghouse approach and made a conclusion with regard to its reasonableness. R. Klecker, NRC Staff, explained that the Westinghouse and NRC Staff are similar except for the Staff's use of RT_{NDT} and betas for description of the temperature drop as opposed to the more severe infinite drop in temperature assumed by Westinghouse.

The Committee explored the Westinghouse approach with D. Speyer. J. Ebersole questioned whether there would be a substantial advantage if the reactor coolant pumps were tripped. D. Speyer indicated there would be a substantial benefit in the heat transfer coefficients but other aspects of the scenario would be much worse.

D. Speyer presented a table of frequency of transients which potentially initiate a crack by class of cooldown transients. It was pointed out from the table that excessive feedwater transients are very benign events with very low probabilities and small break LOCA events are relatively high probability events which tend not to be influenced by operator actions. In answer to a concern by M. Bender, D. Speyer indicated that for small break LOCA events, operator actions were not important since there was automatic actuation of safeguards equipment. Chairman Shewmon questioned why small break LOCA which is postulated to be a high probability event causes pressurized thermal shock. D. Speyer indicated that small break LOCAs result in stagnation in the effected loop.

M. Bender questioned why the excessive feedwater transient probability of thermalized shock was so low. J. Romancick, Westinghouse, indicated that the Westinghouse NSSS design incorporates a number of redundant backup features to isolate feedwater in the event of an excessive cooldown due to a feedwater transient. He indicated also that the inventory in the Westinghouse steam generator is very high, such that it takes a significant amount of water before a substantial amount of cooldown is felt by the system.

J. Ebersole pointed out that after a transient and subsequent safe shutdown, even though there is no actual significant damage, there may be severe monetary impact resulting from detailed fracture mechanics calculations to determine the potential for crack initiation and a decision as to whether inspection is necessary.

D. Speyer presented some recommendations for future research and development efforts in the area of pressurized thermal shock. He discussed fuel management techniques one of which could reduce fluence at the vessel wall. D. Speyer pointed out that in the area of human factors, operator performance could be enhanced through improved procedures and training and quantitative guidance. In answer to a question by Chairman Shewmon, D. Speyer indicated that the temperature limit methodology that Westinghouse is using is being factored into the status trees and function restoration guidelines that the operator will see. He also indicated that despite the appearance of two different approaches, the NRC Staff and Westinghouse are in close agreement on numbers.

D. Peck, Senior Consulting Engineer for Combustion Engineering, summarized the CE approach to pressurized thermal shock which was founded on two basic premises:

- No concern for newer vessels due to low copper materials
- No near-term concern on older vessels.

D. Peck indicated that CE is working on Emergency Procedure Guidelines as part of its post-TMI effort, although no guidelines were found that would cause a pressurized thermal shock event. He indicated that some improvements on the guidelines specifically aimed at pressurized thermal shock will be included in the next revision which will be submitted to the Staff in July, 1982. D. Peck indicated that CE had evaluated different kinds of transient scenarios using linear-elastic fracture mechanics analysis. He indicated that these analyses did not eliminate the preexistence of cracks but use acceptance criteria of crack arrest if there is crack initiation. D. Peck showed a summary of pressurized thermal shock evaluations (see Appendix XVIII). He explained that some transients scenario results include credit for warm prestress. He did point out that the main steam line break and anticipated operating occurrence results did not depend on warm prestress.

He did point out that the main steam line break and anticipated operating occurrence results did not depend on warm prestress. D. Peck also pointed out that the most challenging and most limiting transient for CE plants was the main steam line break.

The Committee discussed the assumptions used in the CE analysis, including reactor coolant pump trip, stagnation and the issue of depressurizations/repressurization. D. P made reference to vessel fluence reduction by fuel management and suggestions for future research and development on the subject of pressurized thermal shock (see Appendix XVIII).

J. Gasper, Manager of Reactor Fuel Technical Services at Omaha Power, presented a discussion of fluence reduction work at the Fort Calhoun Station and excess feedwater events at Rancho Seco and how they might relate to CE plants. He expressed confidence that the main steam line break is the bounding transient for Fort Calhoun, that CE reactors are based on the severity of this thermal transient, and the fact that no higher probability events of higher severity have been found. He expressed confidence that elastic fracture mechanics will probably always show that the vessel will not suffer a through wall crack. At Fort Calhoun he indicated that operator action can be shown to minimize the potential for repressurization. He indicated that training to preclude this event has been completed.

J. Gasper divided his specific comments on neutron flux reduction at the vessel wall into three categories: practical fuel management changes; potential fuel management changes; and reactor vessel wall shielding (see Appendix XIX).

J. Gasper indicated that if an excess feedwater transient were to occur at Fort Calhoun, there would not be dependence on operator action to terminate this type of event. He indicated that the large hot water inventory in the steam generator combined with control and safety systems would give the operator about 20 to 30 minutes to take action. W. Kerr suggested that control system failure might cause the excess feedwater transient. J. Herbst, CE, indicated that the reliability of the main feedwater isolation system is typically of the order of 10^{-3}. The Committee discussed the reliability of the main feedwater isolation system. M. Bender requested that J. Herbst furnish the Committee with information regarding the probability of the availability of the power supply for the main steam isolation valves. J. Ebersole questioned why there was a difference in the probability of an excess feedwater transient in the CE experience than in the Westinghouse work.

J. Gasper indicated that in the CE excess feedwater transient, the minimum temperature was around 440° F which is a lot more probable. He added that if CE were to take the low temperature postulated by Westinghouse, it would yield an extremely low probability also.

J. Ebersole questioned whether there was a need for instrumentation to inform the operator that he has a mismatch in pressure and temperature or cold water in a highly pressured vessel. J. Gasper indicated that CE will provide such a signal with alarm functions pointing to minimum cooling or subcooling temperatures or the operators exceeding the 100° F temperature.

B. J. Short, Project Manager for the B&W Owners Group Program, explained why B&W's approach to the pressurized thermal shock problem is correct (see Appendix XX). He pointed to plant specific analyses conducted which used crack arrest as an acceptance criteria and the same linear elastic fracture mecnanics techniques used by Westinghouse and CE. He indicated that the results of these analyses showed that B&W reactor vessels are acceptable for their remaining lifetimes. M. Bender questioned whether B&W is taking credit for mixing in the downcomer. B. J. Short indicated that B&W considers mixing important and B&W is taking credit for it. B. Short discussed B&W efforts to reduce fluence or flux reduction and human action dependence to avoid a Rancho Seco type transient. J. J. Ray questioned whether changes had been made in the integrated control system (ICS) or the nonnuclear instrumentation systems voluntarily because B&W thought they were necessary after TMI-2. J. Taylor of B&W indicated that desensitization issues such as auxiliary feedwater flow control, auxiliary feedwater activation, and power supplies to the integrated control system were addressed but actual changes to the ICS were not made. He added that most of the changes were identified by the combination of B&W and the utilities with B&W plants. Work that was done by the Staff after the Crystal River event that suggested safety grade auxiliary feedwater and redundant power supplies has already been considered in some plants under construction.

L. Chano, Manager of the Division of Planning at GPU-Nuclear and Chairman of the B&W Owners Group Subcommittee on Materials, presented the GPU-Nuclear approach to pressurized thermal snock (see Appendix XXI). He indicated that a three dimensional mixing process was used and evaluated using the COMEX-1A computer code. He indicated that the analyses showed that there was indeed mixing in the downcomer in B&W vessels.

L. Chano indicated that GPU-Nuclear has an active material surveillance program which has generated accurate copper and phosphorous contents in the vessel wells. L. Chano pointed to a low leakage fuel management scheme being considered by GPU-Nuclear and several plant modifications done to avoid the Rancho Seco type of accident.

B. Hill, Licensing Engineer for Oconee, explained that Duke Power supports the vessel generic effort begun back in 1979. He indicated that a realization at Duke Power that a plant specific analysis was required for more realistic results, resulted in a Duke Power report issued January 2, 1981 using realistic material properties, fluence levels, and assessment of operating experience. He also explained that Duke Power supports the research and development effort concentrated in the areas of review of operating experience. This is so that nothing could potentially happen in the control room that might be a precursor to a transient. He also endorsed a material surveillance program and enhanced inservice examination of the vessel. B. Hill pointed to an 18 month fuel cycle that had been implemented on one unit and the transition taking place on one or two of the other units which involves a lowering of fluence levels. He also mentioned several improvements made at Oconee with regard to the discussion on the Ranch Seco type transient.

D. <u>Comments by ACRS Consultants</u>

T. Theofanous pointed out that the COMIX computer code suffers from a basic flaw because it uses a characterization of laminar diffusion which predicts complete mixing via a numerical diffusion technique. He indicated that the Staff's presentation of the design basis transient is based upon a number of calculations and results which have not been adequately detailed. He suggested that the ACRS should review the details and assumptions in these calculations. T. Theofanous also felt that the cooldown represented by the Staff seemed to be too fast.

F. Binford suggested that reactor operators need diagnostic assistance to cope with these transients, and procedures and training should be properly interfaced with the equipment the operator will have at his disposal. He also felt that the probabilistic approach being used in these analyses should be standardized so that the different vendor approaches could be more easily reconciled.

Z. Zudans suggested that the projection by Combustion Engineering would not be as optimistic had they not assumed that mixing was taking place prior to reaching the vessel downcomer. D. A. Peck of Combustion Engineering pointed out that Combustion Engineering systems generally have low head High Pressure Safety Injection (HPSI), such that when the system repressurizes the amount of HPSI water decreases. Therefore, he indicated, the addition of HPSI water is not a primary cooldown phenomenon. Z. Zudans suggested that this aspect should be explained in greater detail.

J. Ebersole questioned whether the Staff felt it was important to give the reactor operator the total perspective of the actions that would not been adequately detailed. He suggested that the ACRS should review the details and assumptions in these calculations. T. Theofanous also felt that the cooldown represented by the Staff seemed to be too fast.

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J. Ebersole questioned whether the Staff felt it was important to give the reactor operator the total perspective of the actions that would result should he act in one way or another. H. Denton agreed that examination of this point will be a priority item as the Staff visits the plants and examines their training program. H. Denton made some additioral comments:

- There is reasonable agreement between metallurgists how to calculate fracture toughness parameters
- There is disagreement on the probabilities of various transients that actually cool the vessel and how to treat operator actions

- Once the transient lower temperature gets to the temperature of the vessel metal, temperature becomes an extremely important parameter.

M. Bender suggested that the NRC Staff had not given enough weight to treatment of the influence of material properties, and has dealt with them in an arbitrary and very conservative manner.

IV. Meeting with the NRC Commissioners (Open to Public)

[Note: R. F. Fraley was the Designated Federal Employee for this portion of the meeting.]

A. Thermal Shock of Reactor Pressure Vessel

M. Bender explained that the Staff seems to be developing a regulatory framework based on probabilistic arguments which has some validity but also has the usual problems associated with probabilistic arguments. He added that the plan appears to require a considerable amount of investigative experimental work and dialogue with industry. M. Bender expressed concern that the Staff does not appear to have a full understanding of the physical problem with regard to pressurized thermal shock. Chairman Palladino requested a further explanation of that statement as to why he felt that the Staff did not understand the problem. M. Bender indicated that the Staff is interpreting virtually every aspect of the problem in a most conservative way and, while the materials probably have a lot of reserve capability, it is not being credited by the Staff approach.

P. G. Shewmon suggested that the Staff's assumptions, which are not very clear, lead to a bounding "guesstimate" which seems arbitrary and certainly conservative. C. P. Siess suggested that there would certainly be merit in having industry representatives and the Staff confer since industry has also gone through calculations to get bounding cooling curves, different curves that use different assumptions than the Staff. Commissioner Gilinsky suggested that protection in the reactor against a large break in the pressure vessel is one area in which there should be a healthy safety margin.

P. G. Shewmon suggested that one of the critical questions with regard to this problem involves operator actions. Chairman Palladino questioned whether the Committee had any comments with regard to flow mixing problems involved. M. Bender indicated that the bounding computations were likely to signal which vessels are worrisome. He did point out that it should be decided whether it is justified to take some credit for mixing in the case of the high pressure injection system. It might be a legitimate conservatism that has not been credited. M. Bender did suggest that the Staff expand its dialogue with reactor operators concerning the importance of emergency procedures to a pressurized thermal shock transient.

H. Etherington pointed out that it is well understood that there is really no danger of a crack propagating in a cooldown unless the reactor is repressurized cold. He pointed out that the concensus of the ACRS subcommittee was that there may ultimately be a hazard but no real problem for the next few years. He noted, however, that the industry position that there is no problem for the full vessel lifetimes, is not subscribed to by the Staff. Chairman Palladino requested comments and thoughts from the Committee regarding a proposal by Congressman Markey for an in situ annealing demonstration program on an old, embrittled reactor vessel not currently in service.

B. <u>Quantitative Safety Goals</u>

D. Okrent previewed the ACRS' letter on Quantitative Safety Goals by indicating that the Committee will certainly emphasize that the Commission should not take any final action on a policy statement on safety goals until there is a proposed implementation plan that has been reviewed. He added that qualitative goals with quantitative guidance are useful but guidance to the public should be different from that given to the Staff and industry with the Staff having more design oriented goals. In addition, the probabilistic risk assessment which forms one of the bases for the quantitative safety goals should be treated with care because of the large uncertainties that exists in PRA, and the differences that are likely to appear from analyses by different groups for the same system or same plant.

Commission Ahearne questioned whether the ACRS would recommend any changes or suggestions for modifications in the quantitative goals. D. Okrent indicated that the Committee will comment in favor of some criterion on containment as well as on core melt or the prevention of core melt.

Chairman Palladino questioned whether the Committee would comment on the ALARA aspect. He summed up his position regarding ALARA as follows:

- Make efforts to bring operating plants down to an acceptable level of risk

- With regard to plants under design, go to the limit with cost beneficial modifications to meet ALARA.

D. Okrent agreed with the Chairman's concepts. Chairman Palladino and Commissioner Ahearne agreed upon the need for a proposed implementation plan for the safety goals and agreed that the Staff should move on development of such a plan.

C. Proposed ACRS Review of the Clinch River Breeder Reactor (CRBR)

M. W. Carbon pointed out that the review of the CRBR is quite different from a standard review of a light water reactor in that the ACRS is starting even with the Staff in the review and has had to work heavily with the applicant, the Department of Energy. He indicated that the ACRS is working in concert with the Staff but the Staff does not have positions on various technical matters. Chairman Palladino inquired as to particular key points on the ACRS review schedule with respect to the CRBR. M. W. Carbon indicated that the ACRS will review the site suitability at the July meeting and he added that the Commission may expect a construction permit letter in May 1983.

D. Reactor Pressure Vessel Liquid Level/Inventory Instrumentation

W. Kerr reviewed for the Commission the history of this issue starting shortly after the TMI-2 accident with recommendation for installation on operating reactors of an unambiguous water inventory instrumentation system. Palladino inquired whether the ACRS had evaluated the B&W proposal on inventory control. W. Kerr indicated that the B&W proposal was a concept that might be developed into a system but had not yet been developed into a system which could be reviewed.

In answer to a question by Chairman Palladino, W. Kerr and D. Okrent both indicated that the B&W proposal will require further study and additional explanatory details before a judgment can be made as to whether they are as far along with their concept as the other vendors.

In response to a question by Commissioner Ahearne, H. W. Lewis indicated that it was his personal view that the Commission has not yet decided what they want to measure and in what context they want to measure it. He suggested that a better water level indicator might have resulted from a more organized study of the real purpose of the instrumentation. He noted that the pumps on/pumps off issue has not been resolved. He did commend the Staff, however, for taking a more systematic and rational approach to the problem than it had done earlier. In answer to a question by Chairman Palladino, H. W. Lewis indicated that he no longer believed that these systems are counterproductive to safety. However, he suggested that the Commission put a greater premium on reactor operator training and judgment.

E. Proposed NRC Long-Range Research Program Plan

C. P. Siess explained that under existing procedures the ACRS is to review the Long-Range Plan at the draft stage along with user offices. He indicated that it was not clear what the result of the review was intended to be - should it be an input to the Research Staff or comments to the Commission. He explained that ACRS reports on the Research Program are collegial and a concensus of the full Committee, a process involving about ten ACRS subcommittees. It takes at least one full Committee review and close to three months to complete. He indicated that the ACRS plans to meet with the Staff in August to discuss the scope and format of the document. But, the ACRS would just as soon not be in the review process at the draft stage as far as review approval or input to the document. C. P. Siess indicated that the ACRS plans to continue to give the Staff input to its research program in the form of comments and recommendations whether indirectly through meetings or reports to the Commission as well as reports to the Congress. However, he added, the ACRS wishes to make the distinction between review of the Research Program which is almost a continuing effort and ACRS review of a particular NUREG on the Long-Range Research Plan. Chairman Palladino suggested that the ACRS write a letter regarding the basis for proposing that the ACRS not review the Long-Range Research Program Plan at the draft stage for consideration by the Commission.

F. Additional Discussion Issues

Commissioner Ahearne noted the ACRS comment in its letter concerning the draft of a proposed rule for environmental qualification of electrical equipment with respect to the fragmentation of the rule because seismic qualification was not treated. He requested further explanation on that comment. J. J. Ray indicated that it was in the nature of an alert because it left the utility or user with an incomplete picture of the environmental qualification issue, and especially as to whether qualification now for environmental purposes would require requalification later for seismic conditions with the potential for removal of expensive equipment.

In answer to a question by Commissioner Gilinsky as to what the Commission should do about this issue, J. J. Ray indicated that the Commission should proceed with the environmental qualification first and then with specifications for the development of the seismic requirements at a later date. The objective would be to have plants that have not finalized their environmental qualification have the benefit of the newly defined seismic requirements and perhaps do both qualification requirements in close enough timing so that they would not be required to replace equipment for seismic reasons that has been environmentally qualified.

M. Bender pointed out that the seismic qualification has to do with certain hardware that has to be physically oriented in the plant. The mounting arrangement has to be understood as well as just testing the hardware. He indicated that he had been concerned that some equipment would still be open to question with regard to seismic response after other environmental qualification had been done. Since it is anticipated that some seismic qualification is associated with the environmental qualifications, the lack of formal specifications for seismic qualification should not present as serious a matter as once expected.

Commissioner Ahearne questioned whether the ACRS had recommended in its report on SECY-82-111 that the Safety Parameter Display System (SPDS) be safety grade. W. Kerr explained that the comment in the letter was not specifically that the SPDS be safety grade, but that more thought be given to appropriate reliability requirements that should be used with the SPDS. Chairman Shewmon suggested that it might not be a good idea to mandate a requirement for safety grade in that it might hinder the actual development of the instrumentation by industry.

Commissioner Ahearne questioned the activities the ACRS had underway with regard to the area of high-level waste management. D. W. Moeller pointed to a June 8, 1982 subcommittee meeting review of the current status of DOE plans as well as NRC Staff efforts in this area.

H. W. Lewis presented additional views on the Quantitative Safety Goals. He referred to the question "How safe is safe enough", the use of risk aversion, ALARA, and the most exposed individual in the safety goal concept. He suggested that he would much rather see a completely arbitrary overall safety goal, rather than the methodology currently suggested. Chairman Palladino indicated that he had a problem with an arbitrary number because it would have to have some reference point in support of the goals proposed. He indicated that an advantage of this premise from which NRC is starting is that there is at least a reference point given by the probability values in the safety goal.

V. Consideration of Seismic Events in Emergency Planning

[Note: H. Alderman was the Designated Federal Employee for this portion of the meeting.]

D. W. Moeller reported the results from the Reactor Radiological Effects Subcommittee Meeting on May 14, 1982. He indicated that the primary result of small earthquakes on emergency planning would focus on the disruption of roads in the vicinity of a nuclear power plant. He indicated that in a discussion with the NRC Staff, the Staff indicated that backup communication systems and helicopters would be of value in the event of a small earthquake. A question came up as to whether similar preparations should be made with regard to a major earthquake. B. Grimes, NRC Staff, indicated that it would not be appropriate to take these measures with large earthquakes because of the massive nature of the disruption caused. D. W. Moeller indicated that B. Grimes had referred to a misinterpretation made by the Atomic Safety and Licensing Board which was under the impression that the Commission had referred to all earthquakes when discussing seismic events and emergency planning. B. Grimes comments were stated as follows:

- With regard to a small earthquake, the power plant would remain intact and emergency planning for a small earthquake would be beneficial.
- In the event of an intermediate earthquake, evacuation might not be possible and a suggestion might be made for the population to seek shelter.
- In the event of a major earthquake, little could be done offsite to help the indigenous population. Even if the plant survived, there would be no demand for electricity.

D. W. Moeller indicated that the NRC Staff is developing a position paper with regard to consideration of seismic events and emergency planning at nuclear power plants. He suggested that the Committee wait for issuance of the paper and review the draft at that time.

VI. Control Room Habitability in Nuclear Plants (Open to Public)

[H. Alderman was the Designated Federal Employee for this portion of the meeting.]

D. W. Moeller suggested that the ACRS full committee request that the NRC Staff conduct a two hour briefing at the July or August full committee meeting regarding control room habitability, with presentations made by the NRC Staff and possibly architect/engineering firms, consulting firms and the Institute for Nuclear Power Operation. He indicated that problems with regard to control room habitability have been pointed out through Licensee Event Reports (LERs). He indicated that outside consulting firms have been called in to look at control room habitability at certain plants and have pointed out many deficiencies which did not violate the plant's technical specifications. He pointed out that operators cannot inhabit the control room if dampers are set as often designed. Also noted was the fact that control room operators often lack confidence in control room ventilation systems as presently designed. Chairman Shewman recommended that the issue of control room habitability be put on the agenda for the August full committee meeting.

VII. ACRS Responses to Questions from the Subcommittee on Energy Research and Production (Open to Public)

The Committee briefly discussed a draft of responses to questions informally submitted to the ACRS subsequent to testimony given by C. P. Siess regarding the NRC Safety Research Program before the Subcommittee on Energy Research and Production of the U.S. House of Representatives on Science and Technology on May 18, 1982 (see Appendix XXII). The document was referred back for ACRS Staff revision for later consideration during the Meeting.

VIII. Executive Sessions (Open to Public)

[Note: Raymond F. Fraley was the Designated Federal Employee for this portion of the meeting.]

A. ACRS Reports, Letters, and Memoranda

1. ACRS Interim Report on Midland Plant, Units 1 and 2

The Committee prepared a report to the Commissioners of its review of the Midland Plant Units 1 and 2 regarding the request for an operating license. The Committee concluded that, if due regard is given to comments in the body of the report, and subject to satisfactory completion of construction and staffing, operation at power levels up to 5 percent of full power is acceptable. ACRS recommendation regarding operation at full power has been deferred until the Committee has had the opportunity to review the plan for an audit of plant quality and the proposed resolution of the question of natural circulation in the presence of a small break LOCA.

2. ACRS Report on Pressurized Thermal Shock

The Committee prepared a report to the Commissioners of its review of the current status of the pressurized thermal shock problem.

The ACRS noted lack of sufficient information to evaluate the adequacy of an approach by the NRC Staff to develop a regulation based upon a combination of deterministic and probabilistic analyses.

3. ACRS Comments on Proposed Policy Statement on Safety Goals for Nuclear Power Plants (NUREG-0880, "A discussion Paper")

The Committee prepared a report to the Commissioners of its review of NUREG-0880, A Discussion Paper, recommending that final action on adoption of a policy statement on safety goals should be contingent upon proper evaluation and agreement on the implementation plan. The ACRS plans to provide further comments to the Commission after reviewing the Staff plan for implementation. M. Bender and H. W. Lewis appended additional comments. In the body of the ACRS report will also be found responses to the four questions raised by the Commission.

4. ACRS Review of the NRC Long-Range Research Plan

The Committee prepared a report to the Commissioners regarding termination of a formal ACRS report to the Commission on proposed Long-Range Research Plans. The ACRS expects to continue to receive the LRRP, both in draft and final form, and expects to utilize it in its review of and report on the NRC Safety Research Program and Budget for the Commission and the Congress.

5. Response to Commissioner Gilinsky Regarding Seismic Design Suggestions by Professor Paul Jennings

The Committee prepared a report to Commissioner Gilinsky recommending that the suggestions by Professor Paul Jennings on seismic design be considered within the context of a broad review of the NRC Staff's current seismic design practices including the NRC Staff's reassessment of Appendix A to 10 CFR 100. The ACRS suggested that Professor Jennings be invited to participate in this review.

6. ACRS Responses to Questions from the Subcommittee on Energy Research and Production

The Committee endorsed a response to questions received from the Staff of the Subcommittee on Energy Research and Production of the U.S. House of Representatives on Science and Technology with a one week grace period for comments by members.

B. Future Schedule

1. Future Agenda

The Committee agreed on a tentative agenda for the 267th ACRS Meeting, July 8-10, 1982 (see Appendix II).

2. Future Subcommittee Activities

A schedule of future subcommittee activities was distributed to Members (see Appendix III).

C. Nominations for New ACRS Member

The Committee discussed the qualifications of several prospective nominees to replace ACRS Member W. M. Mathis who is retiring and decided to invite the two leading candidates to the August full Committee Meeting to meet with ACRS Members.

D. Review of ICRP-26 and Proposed Changes to 10 CFR 20

D. W. Moeller has been asked to serve on the DOE Headquarters Ad Hoc Committee which will be reviewing ICRP-26/30 and the proposed changes to 10 CFR 20 in the context of practical operational problems envisioned by DOE. The Committee discussed the matter and decided that he should attend as an ACRS observer.

E. Participation in American Nuclear Society (ANS) Panels

W. Kerr has been invited to participate on a panel discussing the subject of degraded reactor cores at the ANS Annual Meeting being held in Los Angeles, June 6-10, 1982. D. Okrent indicated that he has also been invited to participate on a panel discussing quantitative safety goals at this same conference. The Committee offered no objection.

The 266th meeting of the Advisory Committee on Reactor Safeguards was adjourned on Saturday, June 5, 1982 at 12:25 p.m.